Designated Original

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Constellation Energy
Generation Group

March 23, 2007

U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

LER 2007-001, Loss of Electrical Generation Results in Plant Trip

The attached Licensee Event Report (LER) 2007-001 is submitted in accordance with 10 CFR 50.73, Licensee Event Report System. There are no new commitments contained in this submittal. Should you have questions regarding the information in this submittal, please contact Mr. Robert Randall at (585) 771-5219 or robert.randall@constellation.com.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

Attachments: (1) LER 2007-001

cc: S. J. Collins, NRC
D. V. Pickett, NRC
Resident Inspector, NRC (Ginna)

Rec'd at DCD
5/11/07

IE22
1/1

Attachment 1

LER 2007-001

Loss of Electrical Generation Results in Plant Trip

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Loss of Electrical Generation Results in Plant Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	27	2007	2007	- 001 -	00	3	23	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Robert Randall, Director of Licensing	(585) 771-5219

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
D	JJ	CON	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On January 27, 2007, at approximately 2040 EST, with the plant in Mode 1, initially at 100% steady state reactor power, an event occurred resulting in an automatic reactor trip. The Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. Following the reactor trip, all safety systems operated as designed. The reactor was stabilized in Mode 3.

The reactor trip was the result of a loss of electrical load transient, which is attributed to a failure in the Main Turbine Electro Hydraulic Control (EHC) system that caused the four high pressure turbine control valves to rapidly close simultaneously. The loss of electrical generation resulted in a heat-up of the reactor coolant system and an actuation of the Reactor Protection System (RPS) on Over-Temperature Differential Temperature.

Corrective action to prevent recurrence is outlined in Section V.B.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PRE-EVENT PLANT CONDITIONS:

On January 27, 2007 the R.E. Ginna Nuclear Power Plant (Ginna) was in Mode 1 at approximately 100% steady state reactor power.

II. DESCRIPTION OF EVENT:

A. EVENT:

On January 27, 2007, at approximately 2040 EST, Ginna experienced a loss of load transient when gross electrical generation output decreased from approximately 607 MWe to less than 100 MWe over an approximate ten second time period. The loss of electrical generation is attributed to a failure in the Main Turbine Electro Hydraulic Control (EHC) system that caused the four high pressure turbine control valves to rapidly close simultaneously. The loss of electrical generation resulted in a heat-up of the reactor coolant system (RCS) and an automatic reactor trip due to an actuation of the Reactor Protection System (RPS) on Over-Temperature Differential Temperature.

The Control Room operators performed the appropriate actions of Emergency Operating Procedure E-0 (Reactor Trip or Safety Injection). The operators then transitioned to Emergency Operating Procedure ES-0.1 (Reactor Trip Response) when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required.

B. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

C. DATES AND APPROXIMATE TIMES OF MAJOR OCCURENCES:

- January 27, 2007, 2040 EST: automatic reactor trip due to an actuation of the Reactor Protection System (RPS) on Over-Temperature Differential Temperature.

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

E. METHOD OF DISCOVERY:

The reactor trip was immediately apparent due to plant response, alarms, and indications in the Control Room.

F. SAFETY SYSTEM RESPONSES:

Both motor driven auxiliary feedwater (AFW) pumps and the turbine driven AFW pump started due to a signal from the anticipated-transient-without-scram (ATWS) mitigation actuation circuitry (AMSAC), and functioned properly. As a result of the RCS heat-up transient, the RCS experienced a pressurization event that caused both Power-Operated Relief Valves (PORVs) to open twice. The peak main steam pressure obtained during the heat-up of the RCS also caused one of the eight main steam safety valves to open momentarily. Additionally, an automatic trip of the A Main Feedwater Pump as the result of a low seal water differential pressure occurred approximately seven minutes after the reactor trip.

III. CAUSE OF EVENT:

The immediate cause of the reactor trip was a Reactor Protection System (RPS) actuation on Over-Temperature Differential Temperature. The RPS actuation was the result of a loss of electrical generation event.

The underlying cause of the loss of electrical generation event was attributed to a failure in the Main Turbine Electro Hydraulic Control (EHC) System. An EHC System electrical card which provides a common output signal to all four of the high pressure turbine control valve servo cards had an edge connector that was spread and temporarily lost contact with the receiver cabinet side of the circuit. The result was a common output signal that went to zero volts, causing the control valves to close.

The design of the EHC System electrical card was a known potential failure mechanism as evidenced by warning placards inside the control panel. Although the placards provided some degree of information to the workers, the station failed to incorporate adequate guidance into the maintenance procedures for inspecting the card edge connections and seating the cards. This card is a single point vulnerability with regards to a turbine/reactor trip.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

IV. ASSESSMENT OF THE SAFETY CONSEQUENCES OF THE EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2) (iv)(A), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section". The automatic reactor trip is an actuation of the Reactor Protection System, and AFW pump starts are actuations of a PWR auxiliary feedwater system.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no operational or safety consequences or implications attributed to the reactor trip because:

- The two reactor trip breakers opened as required.
- All control and shutdown rods inserted as designed.
- The loss of load transient is similar to the loss of load transients described in the Updated Final Safety Analysis Report (UFSAR). The loss of electrical load transients are performed to evaluate RCS over-heating events and their impact on the following design criteria:

  i) minimum DNBR,
  ii) maximum RCS pressure and,
  iii) maximum Main Steam and SG pressure

  The UFSAR transients were examined and compared to the plant response for the actual event. The plant behavior was found to be bounded by the events detailed in the accident analysis. The UFSAR transients were found to be bounding due to a combination of less limiting actual plant conditions and proper operation of plant equipment in responding to the plant trip.

- The momentary opening of the main steam safety valve is bounded by the UFSAR accident analysis event, which assumes operation of the main steam safety valves.
- The Control Room operators were in the process of manually stopping the A Main Feedwater Pump at the time it automatically tripped and the event did not adversely affect the ability of the operators to respond to the plant transient.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

- The Conditional Core Damage Probability (CCDP) of the event was calculated as 8.53E-07.

Based on the above and the review of post trip data and past plant transients, it can be concluded that the plant operated as designed, that the public's health and safety was assured at all times.

V. CORRECTIVE ACTIONS:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- The card edge connector was repaired, and a full inspection of the card was performed. Testing was completed to verify proper operation of the card and the EHC System.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Maintenance procedures will be revised to include criteria and operating experience for card edge connector inspection.
- Instrument and Control Technician initial and continuing training programs will be evaluated for specific content based on this event.
- A single point vulnerability classification and development of mitigating strategies will be completed for all single point vulnerabilities in the EH system.
- The affected EHC System card has since been replaced during a forced outage.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

No other structures, systems, or components failed as result of this event.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar Ginna LER event historical search was conducted which resulted in no similar events.

C. THE ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) COMPONENT FUNCTION IDENTIFIER AND SYSTEM NAME OF EACH COMPONENT OR SYSTEM REFERRED TO IN THIS LER:

COMPONENT	IEEE 803 FUNCTION NUMBER	IEEE 805 SYSTEM IDENTIFICATION
Pump	P	SJ
Pump	P	BA
Valve, Relief	RV	SB
Valve, Relief	RV	AB
Connector	CON	JJ